Dominion Nuclear Connecticut, Inc.
Millstone Power Station
Rope Ferry Road
Waterford, CT 06385

Dominion

OCT 3 2002

Docket No. 50-336
B18764

RE: 10 CFR 50.73(a)(2)(iv)(A)

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Millstone Power Station, Unit No. 2
Licensee Event Report 2002-005-00
Automatic Reactor Trip Due to Low Steam Generator Level During
Power Ascension

This letter forwards Licensee Event Report (LER) 2002-005-00, documenting an event that occurred at Millstone Power Station, Unit No. 2 on August 7, 2002. This LER is being submitted pursuant to 10 CFR 50.73(a)(2)(iv)(A).

There are no regulatory commitments contained within this letter.

Should you have any questions regarding this submittal, please contact Mr. David Dodson at (860) 447-1791 extension 2346.

Very truly yours,

DOMINION NUCLEAR CONNECTICUT, INC.

Stephen P. Sarver, Acting Director
Nuclear Station Operations and Maintenance

Attachment (1): LER 2002-005-00

cc: H. J. Miller, Region I Administrator
R. B. Ennis, NRC Senior Project Manager, Millstone Unit No. 2
Millstone Senior Resident Inspector

IE22

Attachment 1

Millstone Power Station, Unit No. 2

LER 2002-005-00
Automatic Reactor Trip Due to Low Steam Generator Level
During Power Ascension

NRC FORM 366 (7-2001)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER)		Estimated burden per response to comply with this mandatory information collection request 50 hours Reported lessons learned are incorporated into the licensing process and fed back to industry Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U S Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503 If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection
(See reverse for required number of digits/characters for each block)		

FACILITY NAME (1)	DOCKET NUMBER (2)	PAGE (3)
Millstone Power Station - Unit 2	05000336	1 OF 3

TITLE (4)
Automatic Reactor Trip Due to Low Steam Generator Level During Power Ascension

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
08	07	2002	2002	— 05	00	10	03	2002	FACILITY NAME	DOCKET NUMBER 05000

OPERATING MODE (9)	1
POWER LEVEL (10)	055

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply) (11)

	20 2201(b)		20 2203(a)(3)(ii)		50 73(a)(2)(ii)(B)		50.73(a)(2)(ix)(A)
	20.2201(d)		20.2203(a)(4)		50 73(a)(2)(iii)		50.73(a)(2)(x)
	20 2203(a)(1)		50 36(c)(1)(i)(A)	X	50 73(a)(2)(iv)(A)		73 71(a)(4)
	20 2203(a)(2)(i)		50 36(c)(1)(ii)(A)		50 73(a)(2)(v)(A)		73 71(a)(5)
	20.2203(a)(2)(ii)		50.36(c)(2)		50 73(a)(2)(v)(B)		OTHER
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50 73(a)(2)(v)(C)		Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50 73(a)(2)(v)(D)		
	20.2203(a)(2)(v)		50 73(a)(2)(i)(B)		50 73(a)(2)(vii)		
	20.2203(a)(2)(vi)		50 73(a)(2)(i)(C)		50 73(a)(2)(viii)(A)		
	20.2203(a)(3)(i)		50 73(a)(2)(ii)(A)		50 73(a)(2)(viii)(B)		

LICENSEE CONTACT FOR THIS LER (12)

NAME	TELEPHONE NUMBER (Include Area Code)
David W. Dodson, Supervisor-Licensing	860-447-1791

COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT (13)

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	SJ	V	Crane	Y					

SUPPLEMENTAL REPORT EXPECTED (14)			EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR
☐ YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

ABSTRACT (Limit to 1400 spaces, i e , approximately 15 single-spaced typewritten lines) (16)

On August 7, 2002, with the plant operating in Mode 1 and in power ascension at approximately 55 percent power, an automatic reactor trip occurred due to low steam generator level. Post trip procedures were followed and the plant responded normally without any engineered safety feature actuation. Prior to the event, feedwater was being supplied to the steam generators by the "A" steam generator feedwater pump (SGFP). The event took place during the startup sequence of the "B" SGFP. The discharge flow from the SGFPs is fed into a common header which in turn is sent through the No. 1 feedwater heaters and then to the steam generators. The SGFPs are each protected from reverse flow by a check valve, 2-FW-1A and B. Downstream of the check valves are motor operated isolation valves, 2-FW-38A and B. As part of the startup sequence for placing the "B" SGFP in service, the operators opened the motor operated isolation valve, 2-FW-38B. Almost immediately, steam generator level deviation alarms were received. The operators commenced closing 2-FW-38B, but the time for the motor operated valve closure was not sufficient to recover steam generator levels prior to the reactor trip

The cause of the loss of feedwater flow to the steam generators was the failure of the "B" SGFP discharge check valve (2-FW-1B) due to an inadequate anti-rotation weld The root cause is that the organization failed to install adequate anti-rotation welds in the SGFP discharge check valves in 1990.

The SGFP discharge check valves were inspected and repaired with improved anti-rotation welds

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
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NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

1. Event Description

On August 7, 2002, with the plant operating in Mode 1 and in power ascension at approximately 55 percent power, an automatic reactor trip occurred due to low steam generator [SG] level. Post trip procedures were followed and the plant responded normally without any engineered safety feature actuation. Prior to the event, feedwater [SJ] was being supplied to the steam generators by the "A" steam generator feedwater pump (SGFP). The event took place during the startup sequence of the "B" SGFP. The discharge flow from the SGFPs is fed into a common header which in turn is sent through the No 1 feedwater heaters and then to the steam generators The SGFPs are each protected from reverse flow by a check valve [V], 2-FW-1A and B Downstream of the check valves are motor operated isolation valves, 2-FW-38A and B. As part of the startup sequence for placing the "B" SGFP in service, the operators opened the motor operated isolation valve, 2-FW-38B. Almost immediately, steam generator level deviation alarms were received. The operators commenced closing 2-FW-38B, but the time for the motor operated valve closure was not sufficient to recover steam generator levels prior to the reactor trip

The speed of the "B" SGFP went from 4 revolutions per minute (rpm) (turning gear, forward direction), to 0 rpm (indicative of back leakage stopping the pump rotation), to 100 rpm (indicative of reverse flow) The "B" SGFP turning gear had been engaged prior to the reverse rotation occurring The turning gear was severely damaged with a Plant Equipment Operator observing the catastrophic failure of the turning gear motor fan housing. The Plant Equipment Operators also heard a low, cyclic noise coming from the vicinity of the "B" SGFP although the "B" SGFP had not been started

The investigation determined that the "B" SGFP discharge check valve (2-FW-1B) failed to fully seat. When 2-FW-38B was opened, this allowed reverse flow through the "B" SGFP from the "A" SGFP, diverting feedwater flow from the steam generators

The check valves are 18-inch Crane Chapman L973A Pressure Seal Tilting disk check valves. The disc is attached to the valve body via two hinges or "ears". The hinge pins are removable and are held in place by retaining screws which are threaded through the hinge and not threaded in the hinge pin The retaining screws were originally flush mounted set screws held in place by a seal weld (anti-rotation weld) The SGFP discharge check valves, 2-FW-1A&B, were modified in 1990 following the failure of a retaining set screw. These were replaced with stud bolts whose head has two flats Vendor directions were to place a tack weld adjacent to the flats to secure the stud bolt in place. The tack welds were inadequate to retain the stud bolts, and one stud bolt backed out This allowed the pivot pin to fall out, and the check valve became stuck in the open position.

This event is being reported pursuant to 10 CFR 50 73(a)(2)(iv)(A) as an event that resulted in an automatic reactor trip.

2. Cause

Following the reactor trip, an event review team and root cause team were formed. Their investigations concluded that the cause of the loss of feedwater flow to the steam generators was the failure of the "B" SGFP discharge check valve (2-FW-1B) The discharge check valve failed due an inadequate anti-rotation weld The need for an effective anti-rotation weld was identified in 1990, but the corrective actions taken at that time were inadequate to prevent this failure The root cause is that the organization failed to install adequate anti-rotation welds in the SGFP discharge check valves in 1990

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

3 Assessment of Safety Consequences

The purpose of the feedwater system is to provide water to the steam generators for the transfer of thermal energy from the primary side of the steam generators to the secondary side. The feedwater system is not safety related The loss of feedwater flow to the steam generators which occurred due to the failure of check valve 2-FW-1B resulted in a reactor trip on low steam generator level. The reactor trip was uncomplicated and all required safety equipment functioned as expected This event is considered to be of low safety significance.

4 Corrective Action

The SGFP discharge check valves were inspected and repaired with improved anti-rotation welds

The root cause is that the organization failed to install adequate anti-rotation welds in the SGFP discharge check valves in 1990 The station organization and processes have been significantly strengthened since the weld repair occurred in 1990 No additional corrective actions to address the root cause are required. Corrective actions to address contributing causes are being addressed in accordance with the Millstone Corrective Action Program.

5. Previous Occurrences

None

Energy Industry Identification System (EIIS) codes are identified in the text as [XX].